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(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - ~~Core Performance Branch (CPB)~~Reactor Systems Branch (SRXB)[1]

Secondary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]

I. AREAS OF REVIEW

The ~~CPB~~SRXB[3] evaluates the consequences of a control rod ejection accident ~~in the area of physics~~to determine the potential damage caused to the reactor coolant pressure boundary and to determine whether the fuel damage resulting from such an accident could impair cooling water flow.[4] The review covers the possible initial conditions, rod patterns and worths, scram worth as a function of time, adequacy of the various reactivity coefficients, adequacy of the calculational methods, and any core parameters which affect the peak reactor pressure or the probability of fuel rod failure.

The review also examines potential fission product releases resulting from a rod ejection accident. These releases contribute to the source term in analyses for determining compliance with dose limits specified in 10 CFR 100.11.[5]

This review applies to pressurized water reactors (PWRs) only.[6]

Review Interfaces

The SRXB will coordinate other branch evaluations that interface with the overall review of rod ejection accidents, as follows:[7]

1. SRXB reviews the reactivity coefficient curves and control rod worths as part of its primary review responsibility under Standard Review Plan (SRP) Section 4.3.[8]

2. SRXB reviews the~~The~~[9] relevant thermal-hydraulic analyses ~~are reviewed~~as part of its primary review responsibility under SRP Section 4.4.

3. ~~The AEB~~PERB[10] reviews, as part of its ~~secondary~~primary review responsibility~~,~~ ~~described in the appendix to~~under Appendix A to[11] this SRP section, the radiological consequences of a rod ejection accident. The PERB review~~ by using~~uses[12] a source term for dose calculations based on the amount of failed fuel as obtained by ~~CPB~~SRXB[13] from the reactor core analyses under this SRP section.~~ The evaluation finding provided is as indicated in the attached Appendix.~~[14]

4. The Instrumentation & Controls Branch (HICB) reviews the ~~The~~[15] applicant's determination of the reactor trip delay time, i.e., the time elapsed between the instant the sensed parameter reaches the level for which protective action is required and the onset of negative reactivity insertion, ~~is reviewed~~as part of its primary review responsibility under SRP Sections 7.2 and 7.3.

II. ACCEPTANCE CRITERIA

1. ~~CPB~~SRXB[16] acceptance criteria are based on meeting the requirements of General Design Criterion 28 (GDC 28)~~(Ref. 1)~~[17] as it relates to the effects of postulated reactivity accidents that neither result~~ing~~ in damage to the reactor coolant pressure boundary greater than limited local yielding~~,~~ nor~~ causing~~ cause[18] sufficient damage to impair significantly the capacity to cool the core.

   Regulatory positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in Regulatory Guide 1.77.[19]

   Regulatory Guide 1.77 ~~(Ref. 2)~~[20] identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a control rod ejection accident. Specific criteria used by ~~CPB~~SRXB[21] in evaluating the control rod ejection accident are:

   a. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/g~~m~~[22] at any axial location in any fuel rod.

   b. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Boiler and Pressure Vessel[23] Code ~~(Ref. 3)~~.[24]

~~c.~~2. The requirements of 10 CFR 100.11 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.[25] The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling (DNB) condition is an input to the radiological evaluation performed under SRP Section 15.4.8, Appendix A,[26] by ~~AEB~~PERB.[27] The radiological criteria used in the evaluation of control rod ejection accidents (PWRs) are given in Appendix B of Regulatory Guide 1.77 ~~(Ref. 2)~~.[28]

Technical Rationale

The technical rationale for application of these acceptance criteria to reviewing the spectrum of control rod ejection accidents is discussed in the following paragraphs:[29]

1. Compliance with GDC 28 requires that the reactivity control system be designed with appropriate limits on potential reactivity increases to ensure that the effects of a rod ejection accident can neither result in damage to the reactor coolant pressure boundary nor cause sufficient disturbance to impair the capability to cool the core.

   The requirements of GDC 28 apply to this section because the reviewer evaluates the maximum reactor pressure during any portion of the transient corresponding to a rod ejection. ASME Codes provide guidance concerning the acceptability of anticipated accident pressure. The review also examines the extent of fuel damage resulting from a rod ejection accident. Regulatory Guide 1.77 provides guidance concerning acceptability of anticipated core damage.

   Meeting this criterion provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired by a control rod ejection accident.[30]

2. Compliance with 10 CFR 100.11 requires that the exclusion area and the low population zone be defined on the basis of assurances that specified limits will not be exceeded for radiation doses from postulated fission product releases to individuals at the outer boundaries of those regions.

   The requirements of 10 CFR 100.11 apply to this section because rod ejection accidents are included among the potential accidents for which fission product releases are postulated. Review under SRP Section 15.4.8 determines the source term to be used by the reviewer for Appendix A to this section in evaluating compliance with 10 CFR 100.11. Guidance for determining acceptability of the source term is found in Appendix B to Regulatory Guide 1.77.

   Meeting these requirements provides assurance that offsite radiation doses from a PWR rod ejection accident will not exceed the guideline doses specified in 10 CFR 100.11.[31]

III. REVIEW PROCEDURES

The reviewer should be guided by the general considerations in the introductory section to this SRP chapter (i.e., SRP Section 15.0).[32]

1. Review of the applicant's analyses, showing that ~~the first of the~~[33] acceptance ~~criteria~~criterion II.1.a above is met, proceeds as follows:

   a. A spectrum of initial conditions is considered, which must include both zero-power and full-power conditions, at the[34] beginning and end of a reactor fuel cycle (BOC and EOC), to ~~assure~~ensure[35] examination of upper bounds on possible fuel damage. Initial full-power conditions should include the uncertainties in the calorimetric measurement of power.

   b. From the initial conditions of (a) and from control rod patterns, the limiting rod worth is determined. Where confirmation is considered necessary,[36] the reviewer may calculate, as an audit, the worth of limiting rods.

   c. Reactivity coefficient values corresponding to the limiting initial conditions must be used at the beginning of the transient. The reviewer checks the reactivity coefficient curves used by the applicant with those reviewed by the ~~CPB~~SRXB[37] under SRP Section 4.3. The two coefficients of most interest are the Doppler and moderator coefficients. If no three-dimensional space-time calculation is performed, the reactivity feedback must be conservatively weighted to account for the variation in the missing dimension(s).

   d. The reviewer inspects the control rod insertion assumptions which include: trip parameters, trip delay time, rod velocity curve, and differential rod worth. Trip parameters and delay time are reviewed under SRP Section 7.2. Control rod worth is checked by the reviewer for consistency with the review performed under SRP Section 4.3.

   e. The applicant's analytical methods are reviewed. The reviewer may use the results of previous case work~~,~~ if the analytical methods have been previously reviewed and approved by the staff. Otherwise, he must perform a de novo review on this case. Alternatively,[38] an audit of several calculations, using methods considered acceptable to the staff, may be done by the reviewer (or consultants to the staff). The primary concern of the reviewer is how well the elements of the analytical model represent the true three-dimensional problem. Other items checked by the reviewer include feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

   f. Results of the calculations done by procedures described in steps a–e are expressed as values of the radially averaged fuel rod enthalpy (in units of cal/g~~m~~). The reviewer determines that the maximum value does not exceed 280 cal/g~~m~~.

2. Verification of compliance with ~~the second~~[39] acceptance criterion II.1.b is accomplished as follows:

   a. The same procedures considered in steps a–f above are followed.

b. For each accident, the maximum primary system pressure should be calculated by an analytical method acceptable to the staff or, as before, an independent audit calculation is made by the staff. The reviewer checks the results (as obtained by the applicant or the staff) for compliance with the second criterion.

3. To meet acceptance criterion II.2, the~~The~~[40] number of fuel rods experiencing clad failure is determined (for use by the Appendix A reviewer[41] in evaluating the radiological consequences of the rod ejection accident) by the following procedure:

a. The reviewer determines that an acceptable procedure for calculating a departure from nucleate boiling condition during the reactivity excursion has been used. This may be done by referring to previous cases for the same nuclear steam supply system (NSSS) vendor. If no approved technique is available, as might be the case for the first project using a new or substantially revised model, the reviewer must perform a separate detailed review (which may be documented separately in a topical report).

b. The reviewer must determine that the number of rods used in the radiological evaluation is the number of rods calculated to have a departure from nucleate boiling. Departure from nucleate boiling must be calculated in accordance with the criteria reviewed and accepted under SRP Section 4.4. Typically, the criterion defines a departure from nucleate boiling ratio (DNBR) less than 1.30 when DNB correlations such as W-3 (Ref. 4) or BAW-2 (Ref. 5) are used.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[42]

IV. <u>EVALUATION FINDINGS</u>

The reviewer verifies that sufficient information has been provided and ~~his~~that the[43] review supports conclusions of the following type, to be included in the staff's safety evaluation report (SER):[44]

> The staff concludes that the analysis of the rod ejection accidents is acceptable and meets the requirements of General Design Criterion 28. This conclusion is based on the following:
>
> > The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could (1)[45] result in damage to the reactor coolant pressure boundary greater than limited local yielding~~,~~ or (2) cause sufficient damage that would significantly impair the capability to cool the core. The requirements have been met by demonstrating that the regulatory positions of

Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for ~~PWR's~~Pressurized Water Reactors,"[46] are complied with. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/g~~m~~, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten $U0_2$ was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum control rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[47]

V. IMPLEMENTATION

The following section is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP ~~S~~section.[48]

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[49] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[50]

Implementation schedules for conformance to parts of the method described herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."

2. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."[51]

~~2~~3. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

~~3~~4. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

~~4~~5. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

~~5~~6. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)

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SRP Draft Section 15.4.8

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB name and abbreviation	Changed PRB to Reactor Systems Branch (SRXB).
2.	Current SRB name and abbreviation	Changed SRB to Emergency Preparedness and Radiation Protection Branch (PERB).
3.	Current PRB designation	Changed PRB to SRXB.
4.	Editorial revision	Revised to identify specific areas of review.
5.	Editorial revision	Added to state that the areas of review include potential fission product releases. This is significant because GDC 28 requires evaluation of fuel damage only to determine whether coolant flow could be impaired. Fission product release is another area of review.
6.	Editorial revision	Added a sentence stating that this SRP section applies to PWRs only.
7.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW and provided lead-in sentence.
8.	SRP-UDP format item	Added a review interface with SRP Section 4.3 because this section is identified under review procedures 1.c and 1.d.
9.	Editorial revision	Revised to achieve parallel construction.
10.	SRP-UDP format item	Updated review branch abbreviation.
11.	Editorial revision	Revised to indicate that Appendix A to SRP Section 15.4.8 is essentially an independent review and is the primary review responsibility of PERB.
12.	Editorial revision	Broke the sentence into two sentences for clarity.
13.	SRP-UDP format item	Updated review branch abbreviation.
14.	SRP-UDP format item	Deleted redundant information.
15.	Editorial revision	Revised the sentence to make its structure parallel to other review interfaces. Added the review branch designation.
16.	SRP-UDP format item	Updated review branch abbreviation.
17.	SRP-UDP format item	Provided "GDC 28" as abbreviation for "General Design Criterion 28." Deleted unnecessary reference callout.
18.	Editorial revision	Modified sentence for clarity.
19.	Integrated Impact Number 702	Appendix B of Regulatory Guide 1.77 references an outdated standard: ICRP 2 1959.

SRP Draft Section 15.4.8

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
20.	SRP-UDP format item	Deleted unnecessary reference callout.
21.	Current PRB designation	Changed PRB to SRXB.
22.	Editorial revision	The preferred abbreviation for gram is g, not gm (global change for this SRP section).
23.	Editorial revision	Identified the specific ASME code as an aid to reviewer.
24.	SRP-UDP format item	Deleted unnecessary reference callout.
25.	Editorial revision	Entered 10 CFR 100.11 as an acceptance criterion for the radiological evaluation. Regulatory Guide 1.77 specifically cites this regulation under the Regulatory Position. Part 28 addresses the reactor pressure boundary and the obstruction of core cooling, but not fission product release.
26.	Editorial revision	Appendix A is effectively a separate SRP section. Revised to clarify that the radiological evaluation is done under Appendix A.
27.	SRP-UDP format item	Updated review branch abbreviation.
28.	SRP-UDP format item	Deleted unnecessary reference callout.
29.	SRP-UDP format item	Added "Technical Rationale" to ACCEPTANCE CRITERIA and provided lead-in paragraph.
30.	SRP-UDP format item	Provided technical rationale for GDC 28.
31.	SRP-UDP format item	Provided technical rationale for 10 CFR 100.11.
32.	Editorial revision	Added an instruction that SRP Section 15.0 provides general guidance. A review interface with 15.0 was not added.
33.	Editorial revision	Referred to "criterion 1.a" instead "the first of the criteria" for precision.
34.	Editorial revision	Added article, "the," for readability.
35.	Editorial revision	Changed "assure" to "ensure."
36.	Editorial revision	Added punctuation mark for readability.
37.	SRP-UDP format item	Updated review branch abbreviation.
38.	Editorial revision	Corrected punctuation in paragraph for readability.
39.	Editorial revision	Referred to "criterion 1.b" instead of "the second criterion" for precision.
40.	Editorial revision	Identified this procedure to verify that the new second acceptance criterion is met.

Item	Source	Description
41.	Editorial revision	Added wording to clarify that the evaluation of radiological consequences is done under Appendix A rather than under SRP Section 15.4.8.
42.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
43.	SRP-UDP format item	Changed to eliminate gender-specific reference.
44.	Editorial revision	Added abbreviation, SER.
45.	Editorial revision	Numbered the two key clauses for readability. Eliminated unnecessary comma.
46.	Editorial revision	Spelled out pressurized water reactors as per the title of Regulatory Guide 1.77.
47.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
48.	Editorial revision	Changed upper case to lower case.
49.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
50.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
51.	Editorial revision	Inserted an entry for 10 CFR 100.11, which has been added as an acceptance criterion.

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SRP Draft Section 15.4.8

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
702	Revise Regulatory Guide 1.77 to cite the latest version of ICRP 2.	No changes to SRP.